In situ pressure tests were conducted on the tubes with the largest MRPC indications and the results indicate acceptable margins against burst under normal operating and postulated accident conditions. The NRC had a review conducted by an independent contractor of the in situ test method used at Maine Yankee and determined that it provides a reasonable simulation of the hydraulic pressure loads induced during a postulated main steamline break.

Thus, it has been demonstrated that the tubes with the largest indications at Maine Yankee continued to exhibit adequate structural integrity at the time they were found. This finding is attributable to the morphology of the cracks as determined from metallographic examinations of pulled tube specimens from Maine Yankee. This morphology consists of cracks that were not coplanar but rather of short circumferential length and staggered around the circumference over a short axial region with ligaments of material between the cracks. These ligaments add considerably to the strength of the tube, but these ligaments are generally not detectable by the MRPC.

The findings at Maine Yankee nevertheless raised concern that large undetected circumferential cracks could possibly exist at other plants. Therefore, the NRC issued Generic Letter (GL) 95–03, ''Circumferential Cracking of Steam Generator Tubes,'' on April 28, 1995, notifying licensees of the Maine Yankee experience and requesting that they evaluate recent operating experience concerning the detection and sizing of circumferential cracks and the potential applicability of this experience to their plants. On the basis of the results of this evaluation, past inspections and the results thereof, and other relevant factors, licensees were requested to develop a safety assessment justifying continued operation until the next scheduled steam generator tube inspections were to be performed. The generic letter also requested that licensees develop and submit their plans for the next steam generator tube inspection as they pertain to the detection of circumferential cracks. The utilities were required to respond to GL 95–03 within 60 days. By now, the utilities that own the six plants listed in the Petition have responded to GL 95–03 and the responses have been evaluated by the staff.

Based on the utilities' responses to GL 95–03, with the exception of Millstone Unit 2, the CE plants listed in the Petition have been inspected in those areas susceptible to circumferential cracking with improved eddy current inspection probes equally capable as the Point Plus system of detecting circumferential cracking. All tubes with detected cracks have been removed from service. The licensee for Millstone Unit 2 replaced the original CE steam generators during an outage that ended in January 1993. The new steam generators incorporated many new design features that are expected to eliminate or greatly reduce the potential for circumferential tube cracking. These include the use of Inconel 690, a material that has significantly greater resistance to cracking and hydraulic expansion of tubes, which reduces the potential for cracking in the expansion transitions. The limited operational time, improvements in design, and favorable plant operating conditions minimize the potential for the development of circumferential cracking in the Millstone Unit 2 steam generators. Millstone Unit 2 steam generators will continue to be inspected during refueling outages.

The NRC has studied the risk and potential consequences of a range of SGTR events in NUREG–0844, ''NRC Integrated Program for the Resolution of Unresolved Safety Issues A–3, A–4, and A–5 Regarding Steam Generator Tube Integrity.'' The staff estimated the risk contribution due to the potential for single and multiple SGTRs. The study also examined the expected consequences of SGTR scenarios, including beyond-design-basis situations, such as the potential for release as a result of containment bypass because of failed tubes concurrent with a breach of secondary system integrity. A combination of circumstances and conditions is required to produce such simultaneous failures: (1) Main steamline break or other less severe loss of secondary system integrity, (2) the potential that a population of tubes susceptible to rupture exists in a particular steam generator, (3) the potential that operators would not take actions to avoid high differential pressures, and (4) the probability that a large number of tubes would actually fail simultaneously. In the NUREG –0844 assessment, the staff concluded that the probability of simultaneous multiple tube failure was small (approximately 10^{-5}), and that the risk resulting from releases during SGTRs with loss of secondary system integrity was small (about 10^{-7} latent fatalities per reactor year).

III. Conclusion

Based on the fact that (1) adequate steam generator tube inspections have been performed, (2) primary-to-secondary leakage is being monitored on a continuing basis, and (3) the risk of multiple SGTR events is low, I have concluded that an immediate shutdown and Plus Point probe inspection of Maine Yankee, Fort Calhoun Unit 1, Calvert Cliffs Units 1 and 2, St. Lucie Unit 1, and Millstone Unit 2 is not warranted.

The Petitioner's request for action pursuant to 10 CFR 2.206 is denied. As provided in 10 CFR 2.206(c), a copy of the Decision will be filed with the Secretary of the Commission for the Commission's review. This Decision will constitute the final action of the Commission 25 days after issuance unless the Commission, on its own motion, institutes a review of the Decision within that time.

Dated at Rockville, Maryland, this 6th day of December , 1995.

For the Nuclear Regulatory Commission.

William T. Russell,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 95–30176 Filed 12–11–95; 8:45 am]

BILLING CODE 7590–01–P–M

[Docket Nos. 50–325 AND 50–324]

Carolina Power & Light Company; Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The U.S. Nuclear Regulatory Commission (the Commission or NRC) is considering issuance of an amendment to Facility Operating License Nos. DPR–71 and DPR–62 issued to the Carolina Power & Light Company (the licensee) for operation of the Brunswick Steam Electric Plant, Units 1 and 2 (BSEP) located in Southport, North Carolina.

Effective October 26, 1995, the Commission amended its regulations (10 CFR Part 50, Appendix J) to provide a performance-based option for leakage-rate testing of containments of light-water-cooled nuclear plants. The proposed amendment would permit the licensee to implement this performance-based option, which allows leakage testing intervals to be based on system and component testing performance.

The proposed amendment requires the establishment of a ''Primary Containment Leakage Rate Testing Program'' (program) and makes general reference to the NRC guidance utilized by the licensee for development of this program, i.e. Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program''. Regulatory Guide 1.163 addresses the acceptability of industrydeveloped guidance described in Nuclear Energy Institute document NEI 94–01, entitled ''Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J.'' The proposed amendment takes one exception to the guidance in NEI 94–01. Based upon the use of compensatory measures, the exception would allow the use of less accurate flow measuring equipment.

Certain containment leakage testing schedules and details regarding the scope of containment valves and penetrations to be leak-tested will be included in the licensee's program but would be removed by this proposed amendment from the BSEP Technical Specifications. Consistent with NEI 94–01 the proposed amendment relaxes the schedules for performing primary containment air lock leakage surveillance testing and, if the interval for testing of overall containment leakage (Type A testing) has been extended under the program to 10 years, requires inspections for containment integrity during two other refueling outages before the next Type A test as well as immediately prior to that test.

Before issuance of the proposed license amendments, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendments would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed license amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed license amendments revise the Technical Specifications to reflect the adoption of a performance-based containment leakage-testing program. The Nuclear Regulatory Commission has approved the use of a performance-based option for containment leakage testing programs when it amended 10 CFR Part 50, Appendix J (60 FR 49495).

For adoption of the revised regulations, licensees are required to incorporate into their Technical Specifications, by general reference, the NRC regulatory guide or other plant-specific implementing document [used to develop the performance-based leakage-testing program]. A new Administrative Control subsection is being added to the Brunswick Plant Technical Specifications that requires the establishment and maintenance of a Primary Containment Leakage Rate Testing Program. As stated in the Technical Specification, this Primary Containment Leakage Rate Testing Program will conform with NRC Regulatory Guide 1.163, Revision 0, dated September 1995, ''Performance-Based Containment Leak-Rate Testing Program'' by establishing leakage testing intervals based on the criteria in Section 11.0 of NEI 94–01. The Technical Specifications will continue to require performance of a periodic general visual inspection of the containment to ensure early detection of any structural deterioration of the containment system that might occur.

The effect of increasing containment leakage rate testing intervals has been evaluated by the Nuclear Energy Institute using the methodology described in NUREG–1493 [''Performance-Based Containment Leak-Test Program'', September 1995] and historical representative industry leakage rate testing data. The results of this evaluation, as published in NEI 94–01, Revision 0, are that the increased risk corresponding to the extended test interval is small (less than 0.1 percent of total risk) and compares well to the guidance of the NRC's safety goal. Therefore, adoption of performance-based verification of leakage rates for isolation valves, containment penetrations, and the overall containment boundary will provide an equivalent level of safety and does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed license amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. No safety-related equipment, safety function, or plant operations will be altered as a result of the proposed license amendment. The safety objective for the primary containment is stated in 10 CFR 50, Appendix A, ''General Design Criteria for Nuclear Power Plants.'' The safety function of the primary containment will be met since the containment will continue to provide ''an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment * * *'' for postulated accidents. Therefore, the proposed license amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed license amendments do not involve a significant reduction in a margin of safety. As stated above, the Nuclear Regulatory Commission has approved the use of a performance-based option for containment leakage testing programs when it amended 10 CFR Part 50, Appendix J (60 FR 49495). The new Primary Containment Leakage Rate Testing Program will conform with NRC Regulatory Guide 1.163, Revision 0, dated September 1995, ''Performance-Based Containment Leak-Rate Testing Program'' by requiring that leakage testing intervals be established based on the criteria in Section 11.0 of NEI 94–01, Revision 0.

As discussed in Part 1 above, the effect of increasing containment leakage rate testing intervals has been evaluated by the Nuclear Energy Institute using the methodology described in NUREG–1493 and historical representative industry leakage rate testing data. The results of this evaluation, as published in NEI 94–01, Revision 0, are that the increased safety risk corresponding to the extended test intervals is small (less than 0.1 percent of total risk) and compares well to the guidance of the NRC's safety goal. In addition, as demonstrated by risk analyses contained in NUREG–1482 (sic) [NUREG–1493], relaxation of the integrated leak rate test frequency does not significantly increase the probability or consequences of a previously evaluated accident. Integrated leakage rate tests have been demonstrated to be of limited value in detecting significant leakages from penetrations and isolation valves. Therefore, the proposed license amendments adopting a performance-based approach for verification of leakage rates for isolation valves, containment penetrations, and the containment overall will continue to meet the regulatory goal of providing an essentially leak-tight containment boundary, will provide an equivalent level of safety, and do not involve a significant reduction in a margin of safety.

The revised Technical Specifications will continue to maintain the allowable leak rate (L_a) as the Type A test [containment overall leak-rate test] performance criterion. In addition, a requirement to perform a periodic general visual inspection of the containment has been maintained as part of the performance-based leakage testing program.

The revised Technical Specifications will continue to maintain the allowable leak rate (L_a) (sic) [0.6 L_a] as the Type B [containment penetration leak-rate test] and C [containment isolation valve leak-rate test] tests' performance criterion. As supported by the findings of NUREG–1493, the percentage of leakages detected only by integrated leak rate tests is small (only a few percent) and Type B and C leakage tests are capable of detecting more than 97 percent of containment leakages and virtually all such leakages are identified by local leak rate tests (LLRTs) of containment isolation valves.

Thus, the proposed license amendments do not involve a significant reduction in a margin of safety and will continue to ensure the revised Appendix J regulatory goal of ensuring an essentially leak-tight containment boundary.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish in the Federal Register a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this Federal Register notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC.

The filing of requests for hearing and petitions for leave to intervene is discussed below.

By January 11, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403–3297. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) The nature of the petitioner's right under the Act to be made party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1–(800) 248–5100 (in Missouri 1–(800) 342–6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to David B. Matthews, petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to General Counsel, Carolina Power & Light Company, P.O. Box 1551, Raleigh, North Carolina 27602, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

For further details with respect to this action, see the application for amendment dated September 13, 1995, as amended on November 27, 1995, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403–3297.

Dated at Rockville, Maryland, this 5th day of December 1995.

For the Nuclear Regulatory Commission.

David C. Trimble,

Project Manager, Project Directorate II–1, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

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BILLING CODE 7590–01–P

[IA 95–058]

Five Star Products, Inc. and Construction Products Research, Fairfield, CT and H. Nash Babcock, Order

I

Five Star Products, Inc. (FSP), is a company located in Fairfield, Connecticut, and was formerly known as U.S. Grout Corporation. FSP manufactures and sells grout and concrete products to the nuclear industry and has done so for about 20 years. Through a holding company, Mr. Babcock owns FSP and several related businesses, including Construction Products Research, Inc. (CPR), which performs laboratory tests of FSP products. Mr. Babcock is Vice-President of FSP and President of CPR.

II

FSP submitted its grout and concrete products to CPR for testing. Following the tests, CPR issued certifications that it tested FSP products in conformance with certain specifications of the American Society for Testing and Materials. FSP subsequently utilized those certifications as the basis for certifying that its products satisfied Appendix B and customer Purchase Order (PO) requirements. At various times since 1980, FSP has advertised and represented to NRC licensees that its products are manufactured in accordance with the requirements of Appendix B. It has supplied products pursuant to purchase orders requiring FSP to meet the requirements of Appendix B, and 10 CFR Part 21. Licensees who have purchased material from FSP under FSP's certification of quality have used the grout and concrete in safety-related applications and as basic components.

The Nuclear Regulatory Commission (NRC or Commission) issued 10 CFR Part 21 (Part 21) to implement Section 206 of the Energy Reorganization Act of 1974. Part 21 imposes, *inter alia*, evaluation and reporting requirements on directors and responsible officers of firms which supply basic components of any facility or activity which is licensed or otherwise regulated pursuant to the Atomic Energy Act of 1954, as amended, or the Energy Reorganization Act of 1974. Basic components are structures, systems, or parts in which a defect or failure to comply with applicable requirements could create a substantial safety hazard. 10 CFR 21.3(a). Part 21 is implemented in conjunction with Appendix B, which contains the quality assurance (QA) criteria applicable to design, fabrication, construction, and testing of safety-related structures, systems, and components in commercial nuclear power plants. Together, these requirements are intended to assure the safety of safety-related components, materials, and services for nuclear power plants.

Section 206 of the Energy Reorganization Act of 1974 requires directors and responsible officers of firms constructing, owning, operating or supplying the basic components of a facility or activity licensed or regulated by the Atomic Energy Act of 1954, as amended, who obtain information regarding defects in those basic components, or failures of basic components, or of the facility to comply with NRC requirements, to notify the NRC of those defects and failures to comply. Section 206(d) authorizes the Commission to conduct inspections and other enforcement activities necessary to insure compliance with that section. 10 CFR 21.41 and 21.51 implement Section 206(d).

III

The NRC conducts inspections of vendors who supply safety-related components pursuant to Appendix B and who supply basic components pursuant to Part 21. On August 18, 1992, the NRC began an unannounced inspection of FSP, and of its laboratory contractor, CPR, to determine the extent to which FSP supplied basic components to NRC licensees, the adequacy of FSP's QA Program, the adequacy of CPR's testing of FSP products, and the adequacy of FSP products.

Shortly after the inspection began, Mr. Babcock met with the inspection team and questioned the NRC's authority to conduct the inspection. Mr. Babcock was presented with two identical letters from the NRC staff, dated August 13, 1992, each addressed separately to FSP and CPR. The letters outlined the NRC's inspection authority under 10 CFR Part 21, Section 161o of the Atomic Energy Act of 1954, as amended (AEA), and Section 206(d) of the Energy Reorganization Act of 1974, as amended (ERA). Despite this, Mr. Babcock continued to question the NRC's authority and, throughout the inspection, denied the inspectors access to inspect CPR's testing laboratory, which was located in the basement of FSP's Fairfield, Connecticut, headquarters, and access to inspect CPR's laboratory records.

During the inspection of August 18 and 19, 1992, the inspection team reviewed NRC power reactor licensee POs submitted to Five Star in order to determine the scope of FSP's nuclear involvement. The team was provided with POs for the period 1988 to 1992. Those POs demonstrate that at least seven NRC reactor licensees and one licensee contractor had issued POs to FSP for safety-related grout and concrete mix products, and had specified compliance with Appendix B and Part 21.

The inspection team reviewed copies of several NRC licensee audit reports of FSP and CPR. These reports documented that NRC licensee requests to audit CPR's test laboratory and records were consistently denied by FSP. Further, several NRC licensee audit reports found that FSP's QA program was not acceptable and did not meet certain requirements of Appendix B.

The NRC inspection team requested copies of all audits performed by FSP of CPR to determine CPR's compliance with the quality assurance criteria of Appendix B and Part 21. Only one FSP audit of CPR was performed, by the FSP QA Manager, and it was provided to the NRC inspection team by the FSP QA Manager. The July 31, 1992 audit report concluded that CPR's June 10, 1992 QA program was satisfactory. The format and most of the language of this report were identical to a report of an audit conducted by Toledo Edison, an NRC